NON-PUBLIC?: N
ACCESSION #: 9104090308
LICENSEE EVENT REPORT (LER)

FACILITY NAME: Brunswick Steam Electric Plant Unit 1 PAGE: 1 OF 10

DOCKET NUMBER: 05000325

TITLE: OVERCURRENT RELAY SETTING WAS NOT RETURNED TO PROPER CONFIGURATION FOLLOWING CALIBRATION RESULTING IN GENERATOR TRIP/REACTOR SCRAM
EVENT DATE: 03/05/91 LER #: 91-007-00 REPORT DATE: 04/04/91

OTHER FACILITIES INVOLVED: DOCKET NO: 05000

OPERATING MODE: 1 POWER LEVEL: 100

THIS REPORT IS SUBMITTED PURSUANT TO THE REQUIREMENTS OF 10 CFR SECTION:
50.73(a)(2)(iv)

LICENSEE CONTACT FOR THIS LER:
NAME: WILLIAM R. TOLER, REGULATORY TELEPHONE: (919) 457-2701
COMPLIANCE SPECIALIST

COMPONENT FAILURE DESCRIPTION:
CAUSE: SYSTEM: COMPONENT: MANUFACTURER:
REPORTABLE NPRDS:

SUPPLEMENTAL REPORT EXPECTED: NO

ABSTRACT:

On March 5, 1991, the Unit 1 Reactor was at 100% power and 1005 psig. The Emergency Core Cooling Systems (ECCS) were operable, in standby readiness. The Control Operator (CO) had reduced generator voltage to bring MegaVars to +100. At 0937 hours EST, Unit 1 scrammed. The SCRAM was caused by a turbine generator trip resulting from tripping of the 51V-B main generator overcurrent relay from a sensed overcurrent condition. The variation in the amperage seen as a result of the CO lowering the bus voltage triggered the relay to sense and overcurrent condition; however, no actual overcurrent condition existed during this event.

Immediately following the SCRAM, one of the main generator overcurrent relays (51V-B) was observed to have different settings than the two adjacent relays. This relay had been removed from the plant and taken to the Transmission Substation Maintenance (TSM) shop for calibration during the recently completed refueling outage. It was determined this event resulted from personnel error, the failure of the individual(s) performing the calibration and installation of the relay to return the relay settings to the correct operating configuration following calibration. A Human Performance Enhancement System (HPES) evaluation is being performed. The 51V-B relay settings were corrected and the relay returned to service. Relays worked by the involved TSM crew were visually inspected for proper settings on both units 1 and 2. The safety significance of this event is minimal. During the transient, no unusual safety system performance was noted. No ECCS system initiations were seen nor expected. Other SCRAMS in the past two years that have been attributed to personnel error are reported in LERs 2-89-009, 2-90-009, and 2-91-001.

END OF ABSTRACT

EVENT

Unit 1 Turbine trip/Reactor SCRAM as a result of a generator trip on a "B" phase generator overcurrent relay trip.

INITIAL CONDITIONS

On March 5, 1991, Unit 1 was operating at 100% power. Reactor pressure was at 1005 psig, and reactor vessel level was at 192 inches. Reactor Feedwater Level Control was in automatic, three element control. The following systems were operable in standby readiness: High Pressure Coolant Injection (HPCI), Reactor Core Isolation Cooling (RCIC), Automatic Depressurization System (ADS), Reactor Protection System (RPS), Residual Heat Removal (RHR)/Low Pressure Coolant Injection (LPCI), Core Spray (CS), Standby Gas Treatment (SBGT), Standby Liquid Control (SLC), and Emergency Diesel Generator (EDG).

Unit 1 had been synchronized to the grid for seven days following completion of a refuel/recirculation pipe replacement outage, Grid voltage was about 232 kilovolts and the unit was carrying approximately +150 MegaVars.

EVENT DESCRIPTION

In order to reduce MVARS slightly the Unit 1 Control Operator (CO) had discussed with the CP&L system load dispatcher and received concurrence to lower the main generator excitation. The CO reduced generator voltage to bring MVARS to +100. No problems were encountered and generator response was as expected. At 0936, following the load reduction, "A" bus voltage was verified to be at 230.4 kV and "B" Bus voltage was at 231.2 kV. The CO started to contact the dispatcher to request a capacitor bank be put in service to help raise bus voltage and further reduce MVARS; however, prior to the dispatcher answering the phone, Unit 1 scrammed at 0937. The SCRAM was caused by a turbine generator trip resulting from a tripping of the 51V-B main generator overcurrent relay from a sensed overcurrent condition.

Following the SCRAM, Emergency Operating Procedure (EOP)-01 was entered and plant recovery executed. Systems responded as expected. Neither the Emergency Core Cooling Systems (ECCS) nor the Reactor Core Isolation Cooling (RCIC) system were used. Peak reactor pressure during the transient was 1100 psig, and minimum level was 144 inches. Reactor water Low level 1 setpoint was reached, initiating Group 2, 6, and 8 isolation signals. An existing Group 8 isolation signal was present from normal reactor pressure, therefore no valve motion occurred. Group 2 and Group 6 valves isolated as designed. At 0945 the Startup Level Control Valve was placed in service. The Site Incident Investigation Team (SIIT) was convened to begin investigation of the event. A detailed Sequence of Events is provided in Attachment 1.

EVENT INVESTIGATION

Immediately following the SCRAM, one of the main generator overcurrent relays

(51V-B) was observed to have different settings than the two adjacent relays. The setting for the relays are designed to be on the 10 amp tap and 3 on the time lever. The 51V-B relay was found set on the 4 amp tap, with the time lever on 10. This meant that, with a current to transformer ratio of 5000:1, the 4 amp tap would begin to close at 20,000 amps, instead of a normal setting of 50,000 amps. Calculations based on the plant conditions prior to the SCRAM showed that the generator output current was within the operating range of the relay as-found condition. The variation in the amperage seen as a result of the CO lowering the plant bus voltage triggered the relay to sense an overcurrent condition; however, no actual overcurrent condition existed during this event.

ROOT CAUSE DETERMINATION

The relay was removed from service and a calibration check performed.

The calibration check determined that the relay was properly calibrated within the required tolerance for the 4 amp 10 lever as-found condition.

The 51V-B relay was last calibrated during the recently completed refueling outage. Calibrations of the overcurrent relays are performed by the CP&L Wilmington Transmission Substation Maintenance (TSM) crew, using a vendor (General Electric) relay calibration procedure. The relays are removed from the plant and taken to the Transmission Substation Maintenance shop for these calibrations.

The last test in the calibration process verifies the relay operating time by placing the relay in the 4 Tap, 10 Lever configuration. Following this last calibration step the relay was not returned to the application settings of 10 Tap, 3 Lever. The relay was brought back to the plant and put into service in the as-found 4 Tap, 10 Lever configuration.

The TSM crew uses relay cards to identify the design operating configuration that a particular relay is to be left in. For relay 51V-B, the card correctly stated that the relay was to be set in an operating condition of 10 Tap, 3 Lever.

This event was the result of a personnel error, the failure of the individual(s) performing the calibration and installation of the relay to return the relay settings to the correct operating configuration following calibration. A Human Performance Enhancement System (HPES) evaluation is being performed on the personnel error involved in this event. The investigation will determine the causal factors involved in this event and additional corrective actions that may be necessary to prevent recurrence.

ABNORMAL TRANSIENT OCCURRENCES

During the transient, no unusual safety system performance was noted. No ECCS system initiations were seen nor expected. During SCRAM recovery, the following concerns were noted:

1. While attempting to start the 1A Reactor Recirculation Pump the pump had tripped following the SCRAM due to an Anticipated Transient Without Scram (ATWS)/Recirculation Pump Trip (RPT) signal), the 1A Motor Generator (MG)

Set field breaker did not latch. This is a similar occurrence to that seen in the Unit 2 SCRAM on January 25, 1991. Work Request/Job

Order 91-ABFW2 was initiated to investigate this problem.

Investigation of this event determined that a sequence timer un
t on
the initiation circuitry would not initiate timing on every start attempt. The timer unit was replaced, and the MG Set operated successfully.

2. Following the unsuccessful start attempt on the 1A Reactor Recirculation pump, Operations attempted to open the Recirculation pump discharge valve (1-B32-F031A). The valve would not move off the seat. WR/JO 91-AETN1 was written to investigate the problem.

Initial troubleshooting Motor Control Center indicated acceptable motor bridge and megger readings but widely varying current readings on the three phases when attempting to electrically stroke the valve. Dual position indication was not achieved. This indicated a failure of the motor's magnesium rotor.

General Electric SIL # 368 discussed potential valve locking of Recirculation pump discharge isolation valves. Pressure locking of these valves is considered possible when a bolted-bonnet gate valve is closed with the system full and water is trapped in the body/bonnet cavity above the disc ring seals. If the line is then exposed to a temperature transient, the entrapped liquid can be at a higher pressure than the line. This pressure differential can cause forces on the disc/seat ring seals that are high enough that the valve cannot be operated.

Investigation of the Emergency Response Facility Information System data determined that the F031A valve was closed at 10:35 with the loop pressure at 877 psig. An attempt to open the valve was not made until 10:52, 17 minutes following valve minutes valve closure. At this time, loop pressure had decreased to 418 psig. This created the pressure differential necessary to cause the inability of the F031A motor to unseat the valve and the subsequent motor failure.

3. During the event, thermal cycling was noted in the "B" Feedwater line. The thermal occurred as the Startup Level Control Valve (SULCV) was being cycled. The cycling of the SULCV varied the amount of cool feedwater being mixed with the much hotter water from the Reactor Water Cleanup Heat Exchanger return line. Contact was made with the Nuclear Engineering Department (NED) to determine whether the thermal cycling seen during this event would impact the indication (NRC TAC Number 79381) noted in the weld on the 4D

Feedwater Nozzle. NED, utilizing the analysis supplied by Structural Integrity Associates (SIA), determined that the weld area would be protected from the effects of the thermal transients due to the protection afforded by the thermal sleeve. NED and SIA confirmed that the determination was valid for this occurrence.

4. Reactor vessel bottom head temperature indication, as measured on the

bottom head drain line, showed a cooldown of 170 degrees in one hour during the course of the event. Technical Support reviewed the cooldown and the affected areas and determined that the cooldown experienced during this event did not exceed the margin to brittle fracture allowed by Technical Specifications.

IMMEDIATE CORRECTIVE ACTIONS

The 51V-B relay settings were corrected and the relay returned to service. Since no documentation mechanism exists to ensure the relay's proper setting has been restored other than the technicians performing the testing, the following actions were initiated to ensure no other relays were incorrectly set:

1. The Unit 1 relays worked by the TSM crew were visually checked for proper settings, as identified by the TSM relay cards.

2. The Unit 2 relays worked by the TSM crew that could be checked for proper setting without removing covers were visually inspected to verify proper settings, as identified by the TSM relay cards.

In addition, the Plant General Manager requested the Nuclear Assessment Department (NAD) to review specific activities performed by the TSM crew, and to provide recommendations of additional corrective actions that may be necessary to prevent recurrence prior to closing the generator field breaker. The recommendations included the following:

1. The plant and transmission maintenance staffs develop a list of work performed on the offsite electrical supply system during the past outage.

2. A re-check/re-inspection list be developed based on the criticality of the equipment.

An action item was established for the Shift Manager to ensure that the recommendations identified by NAD during their review of Wilmington Transmission Substation Maintenance work were adequately addressed prior to closing the generator field breaker. Technical Support, Plant Maintenance, the Wilmington Substation Maintenance Crew, and the Wilmington Relay Maintenance Crew completed visual inspections on the generator connections, the isophase bus, the non-segrated bus and associated components, transformers, breakers, switches, and relays that had been worked by the Relay Maintenance crew during the past Unit 1 outage, to ensure the integrity of the system. The inspections were completed by 3/7/91, prior to closing the generator field breaker.

To allow for Unit restart, a checklist was developed for the Shift Manager to resolve, incorporating corrective actions deemed necessary by the SIIT and Plant Nuclear Safety Committee (PNSC), including the following items:

1. Evaluate and repair the B32-F031A valve.

2. Restart the 1A MG Set and verify that no problems exist.

3. Resolve the issue with the cooldown of the bottom head drain line.

4. Determine if the thermal cycling seen during this event in the "B" Feedwater line would have any impact on the known indication on Nozzle 4D.

These items were satisfactorily resolved by 3/7/91, prior to restart.

ADDITIONAL CORRECTIVE ACTIONS

In addition to the immediate corrective actions determined necessary by the SIIT and PNSC, additional long term corrective actions were considered by the SIIT as a result of this event. Significant items relative to this event include:

1. The Wilmington Transmission Substation Maintenance unit will perform an HPES evaluation, with support from the Brunswick Technical Support group.

2. Technical Support is to coordinate developing a plan to address the recurring starting problems with the Reactor Recirculation MG Sets.

3. Technical Support is to provide a report on the cause of the failure of the 1-B32-F031A valve, along with recommended corrective actions.

4. Training is to be provided to appropriate Operations personnel on the failure of the 1-B32-F031A valve, in conjunction with a review of thermal binding and pressure locking.

EVENT ASSESSMENT

The Unit was operating at a maximum power level at the time of the event. The Unit response for this event was within the bounding parameters of the corresponding FSAR Chapter 15 Generator trip event. The equipment concerns identified during the event did not significantly hamper operator ability to achieve and maintain the reactor in a shutdown condition. This event would not have been more severe under any other credible and reasonable conditions.

Other SCRAMs in the past two years at Brunswick that have been attributed to personnel errors have been reported in LERs 2-89-09, 2-90-09, and 2-91-01.

This event was initially reported to the NRC under the four hour reporting criteria, 10CFR50.72(b)(2)(ii), an RPS actuation, including plant SCRAM.

EIIS CODES

SYSTEM/COMPONENT CODE

Automatic Depressurization System *
Core Spray BM

Primary Containment Isolation System JM
Reactor Core Isolation Cooling System BN
Residual Heat Removal/Low Pressure Coolant Injection BO
Standby Gas Treatment System BH
Emergency Diesel Generator EK
Reactor Protection System JE
Reactor Water Cleanup System CE
Standby Liquid Control BR
Turbine Stop Valve TA/ISV
Reactor Feed Pump SJ/P
Startup Level Control Valve SD/LCV
Process Computer IO/CPU
Reactor Recirculation Pump RR/P
Emergency Response Facility Information System IQ

Reactor Recirculation Pump Discharge Valve RR/ISO
RHR Heat Exchanger Outlet Valve BO/ISO
Reactor Recirculation Motor-Generator Sets RR/MG
Safety Relief Valve */RV
Main Generator Overcurrent Relay

* No EIIS System Identifier Found

ATTACHMENT 1

SEQUENCE OF EVENTS--MARCH 5, 1991 SCRAM

The following is a SEQUENCE OF EVENTS for the March 5, 1991 SCRAM. The times shown below are referenced to the Process Computer clock. Event times taken from the Emergency Facility Information System (ERFIS) printout have been modified by subtracting 57 seconds.

INITIAL CONDITIONS

Unit 1 had been synchronized to the grid for seven days. The unit was operating in a steady state condition at 100% power. The grid voltage was approximately 232 kilovolts, and Unit 1 was carrying +150 MVARS. In order to lower MVARS slightly, the Control Operator
egan lowering the
main generator excitation by lowering generator output voltage.

TIME

09:36 Control Operator lowers generator output voltage slightly, reducing the reactive component to approximately +100 MVARS.

09:37:21 Generator phase "B" overcurrent relay trips, creating a generator backup lockout command.

A Main Turbine Trip signal is generated.

A command signal is sent to open the generator output breakers.

A RPS Reactor SCRAM signal is generated. Turbine control valve fast closure is the first signal received. The automatic SCRAM command initiates Control Rod insertion. RPS Reactor high pressure condition and trip signal is also received.

09:37:22 Bypass valves are full open.

09:37:35 RPS channels detect Reactor low water condition and trip. Low Reactor water level 1 setpoint creates Group 2, 6, and 8 isolation signals. Both Group 2 and Group 6 trips occur as required. The Group 2 isolation signal also generates a TIP withdrawal command. A pre-existing Group 8 isolation signal was present from normal reactor pressure; therefore, no valve movement occurred as a result of this signal.

09:37:24 Alternate Rod Injection (ARI) initiation signal is generated due to Reactor pressure increase. Both Reactor Recirculation pump MG sets trip from this signal.

ATTACHMENT 1 (CONT.)

09:37:25 Peak Reactor pressure observed is between 1099 and 1100 psig.

09:37:27 Both the inboard and outboard isolation valves for the Drywell Equipment and Floor Drain Systems' lines are full closed due to the Group 2 isolation command.

09:37:28 Minimum Reactor Water level observed between 144 and 146 inches.

09:37:36 Reactor pressure decreases below the SCRAM setpoint and the four channels reset.

09:37:38 Reactor water level recovers above the SCRAM setpoint and the four channels reset.

09:37:39 Operator transfers the Reactor mode switch to "SHUTDOWN", and inserts a manual SCRAM signal.

09:37:40 Reactor recirculation pump speed passes below 25% during coastdown.

09:37:45 Operator trips the 1A Reactor Feed Pump.

09:37:46 Bus 1B, aligned to the UAT, trips on undervoltage.

09:37:48-50 The four RPS channels detect a "HI-HI" level in the SCRAM discharge volume and trip.

09:37:53 Reactor Pressure is under control, and total bypass valve position is less than 4% open.

09:39:23 The four TIP detectors have retracted and the ball valves have closed.

09:45 Startup Level Control Valve is placed in service.

09:48 Operator resets the ARI initiation signal.

09:49 Operator bypasses the HI-HI SCRAM Discharge Volume level and resets the SCRAM signal. A computer scan of rod position confirms all rods are fully inserted.

09:50 Operator resets Group 2 and Group 6 isolation signals.

09:52 Operator resets generator backup lockout signal.

09:54 Mode switch locked in "SHUTDOWN".

ATTACHMENT 1 (CONT.)

09:58 SRM detectors are fully inserted.

10:00-10:02 SCRAM discharge volume Hi-Hi resets on channel A1, A2, and B2.

10:07 SCRAM discharge volume HI-HI signal resets on channel B1.

10:18 Re-energized Bus 1B from the SAT.

10:30 Began reducing vessel pressure using bypass valves.

10:35 Discharge valve for the "A" Reactor Recirculation pump is closed (1-B32-F031A).

10:47 Verified the delta temperature between the steam dome and bottom drain to be less than 140 degrees, in order to restart Reactor Recirculation pumps.

10:49 Attempted to start the 1A Reactor Recirculation pump.
Drive motor started, but the generator field breaker did
not close. Secured the MG set drive motor.

10:50 Discharge valve for the "B" Reactor Recirculation pump is
closed.

10:51 1B Reactor Recirculation is successfully started.
Discharge valve is reopened and forced core flow is
restored.

10:52 Attempted to open the F031A discharge valve, but valve
would not open.

ATTACHMENT 1 TO 9104090308 PAGE 1 OF 1

CP&L
Carolina Power & Light Company

Brunswick Nuclear Project
P. O. Box 10429
Southport, N.C. 28461-0429
April 4, 1991

FILE: B09-13510C 10CFR50.73
SERIAL: BSEP/91-0150

U.S. Nuclear Regulatory Commission
ATTN: Document Control Desk
Washington, D. C. 20555

BRUNSWICK STEAM ELECTRIC PLANT UNIT 1
DOCKET NO. 50-325
LICENSE NO. DPR-71
LICENSEE EVENT REPORT 1-91-007

Gentlemen:

In accordance with Title 10 of the Code of Federal Regulations, the enclosed Licensee Event Report is submitted. This report fulfills the requirement for a written report within thirty (30) days of a reportable occurrence and is submitted in accordance with the format set forth in NUREG-1022, September 1983.

Very truly yours

J. W. Spencer, General Manager
Brunswick Nuclear Project

WRT/
Enclosure

cc: Mr. S. D. Ebneter
Mr. N. B. Le
BSEP NRC Resident Office

*** END OF DOCUMENT ***